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NRC Research On Reactor Internals

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- **Major concern regarding the structural and functional integrity of core internal components**

  1. **IASCC of Austenitic Stainless Steels**
  2. **Addressing Nuclear Plant Aging and License Renewal Issues**

Irradiation and its Effects on Materials - Perspective

1. Significant Increase in Yield Strength
2. Loss of Ductility,
3. Degradation of Fracture Toughness,
4. Susceptibility to Irradiation Assisted Stress Corrosion Cracking (IASCC),
5. Void Swelling, and
6. Radiation Creep Relaxation.

Neutron Irradiation BWRs

1. Changes the Water Chemistry In (Radiolysis).
2. Increase in Corrosion Potential

Testing, Evaluation and Research on Irradiated Stainless Steels

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- **NRC conducts and participates in programs to provide confirmatory research**
  - **Characterize irradiated materials; understand and validate the deformation and fracture criteria**
  - **Conduct anticipatory, or forward-looking research, for license renewal or new reactor applications**
  - **Investigate material performance over licensing period and beyond.**
- **NRC has completed and planned the BWR-related testing and research, and concurrently increasing focus on the PWR-related research**
  - **Acquire additional irradiated materials, especially cast stainless steels, in order to address program requirements**
  - **Zorita is a potentially useful source of additional LWR-irradiated stainless steel to supplement Halden and BOR-60 irradiated materials**

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Threshold and Saturation

Threshold : Lower Boundary condition to observe a specific effect

LOW DOSE / dpa

Saturation : Upper Boundary condition to observe a specific effect

High Dose / dpa

Slow Irradiation : No shock
- **Time for stress induced damage recovery**

Fast Irradiation : Not enough time for any stress recovery before damage

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Primary Interest to the NRC Program

- **Materials series with low dose exposure**
  - **Help understand thresholds for irradiation effects:**
    - ➔ **Fracture and tearing toughness,**
    - ➔ **Irradiation-Assisted Stress Corrosion Cracking**

- **Materials series with high dose exposure**
  - **Help understand if saturation of mechanical properties occurs:**
    - ➔ **Radiation-induced segregation**
    - ➔ **Void swelling (if any at a given temperature and exposure)**
    - ➔ **Fracture and mechanical properties**

Technical Issues Addressed by NRC Research

- ➔ IGSCC – BWR/PWR
- ➔ IASCC-BWR/PWR
- ➔ Void Swelling
- ➔ Radiation Embrittlement
- ➔ Thermal and Radiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Materials
- ➔ Radiation-Induced Stress Relaxation

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Degradation of BWR Internals

- Started years ago to address IGSCC of sensitized stainless steel.
- Conducted under test conditions corresponding to both normal water chemistry and hydrogen water chemistry
  - (fluence to ~3 dpa, high- and low-electrochemical potential test conditions)
- Conducted at ANL, additional technical analyses by NRC
- Testing highly dependent on specimens from Halden irradiations – specimens for several years of testing

➔ Work on materials from Zorita could augment results obtained from Halden irradiations

Degradation of PWR Internals

- Few resources expended to date, PWR effort increasing as BWR effort concludes
- Multifaceted: IASCC, embrittlement, creep, ever-changing compositions, radiation-induced segregation, void swelling, etc.
- Most test specimens from BOR-60 (fast reactor) augmented by irradiations using Halden reactor
- Track international results to capture all important information
- Current tasks expected to produce data needed for license renewal arena
  - Zorita core support materials would augment & corroborate fast reactor irradiations

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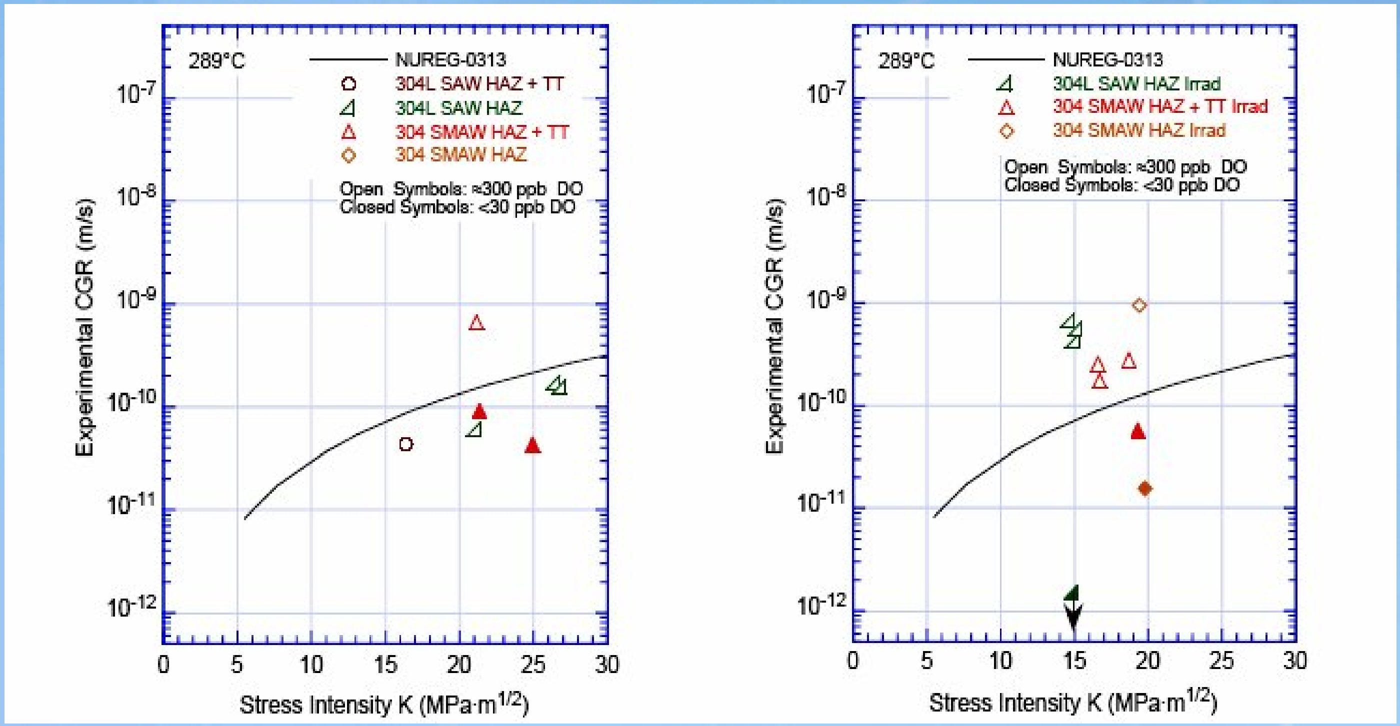
RES Programs in Irradiation-Assisted Stress Corrosion Cracking and Irradiation-Induced Degradation

- ➔ Irradiation-assisted stress corrosion cracking (IASCC) program .
  - – growth rates and fracture toughness in several types of wrought & cast stainless steel
- ➔ Micro-structural studies of radiation-induced segregation, effects of materials chemistry & void development on mechanical properties.
- ➔ Emphasis has been on BWR-applicable fluence levels and coolant chemistries and is transitioning to PWR-applicable testing

IASCC Results for Stainless Steel HAZs, Non-irradiated and Irradiated (to 0.75 dpa)

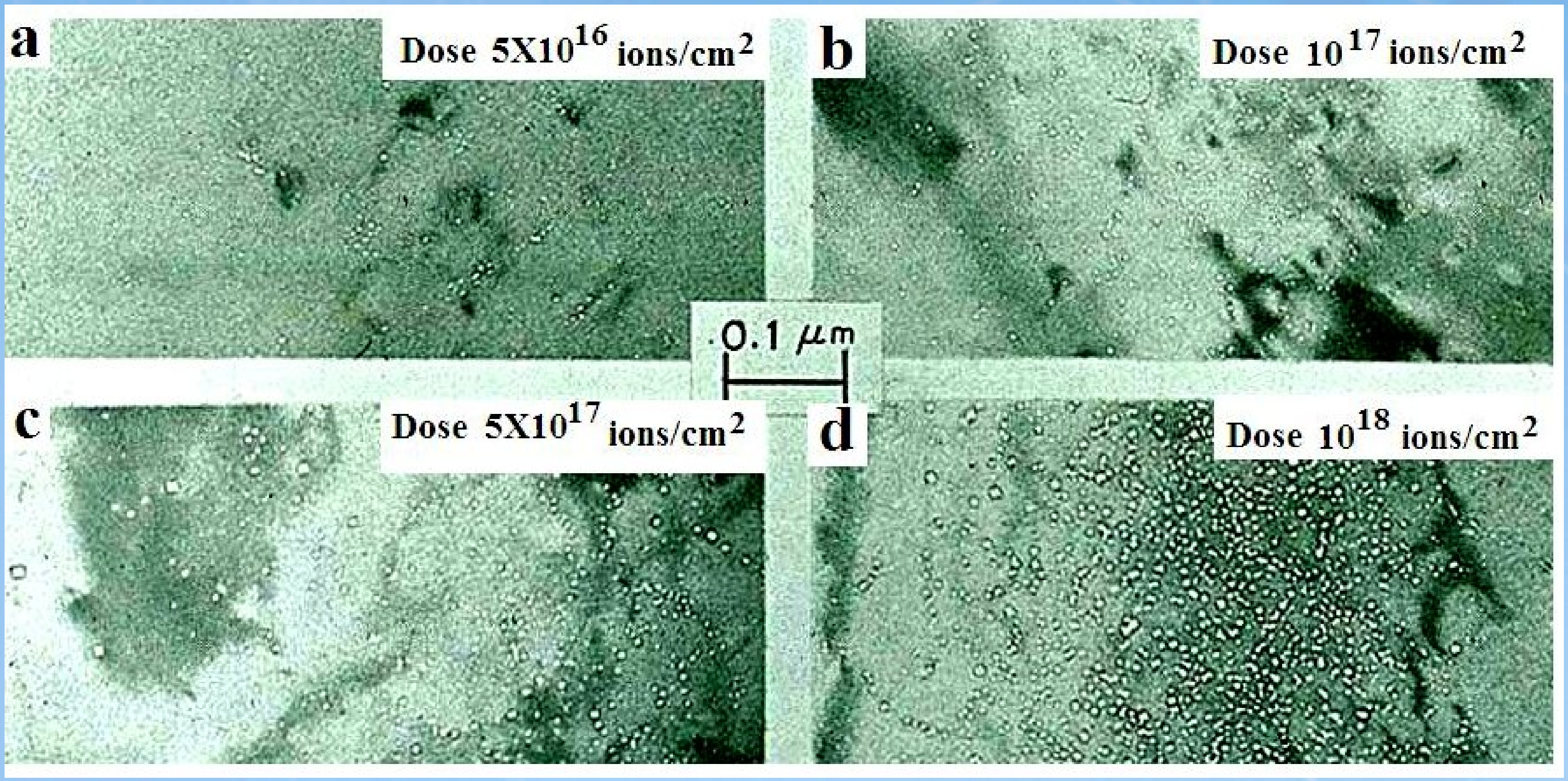
Non-irradiated SS **Irradiated SS**



TT = thermally treated at 500C for 24 hours to simulate low-temperature sensitization & 300 ppb DO represents NWC, 30 ppb DO represents HWC

Void Swelling

- Void swelling of most reactors internals is not expected to be limiting over the current licensing period
- Continued research work will explore the extent of void swelling over the extended operating life of present reactors (i.e 54 EFPY)



Helium bubbles in Mono crystalline Nickel (110) Irradiated with 100 keV ions

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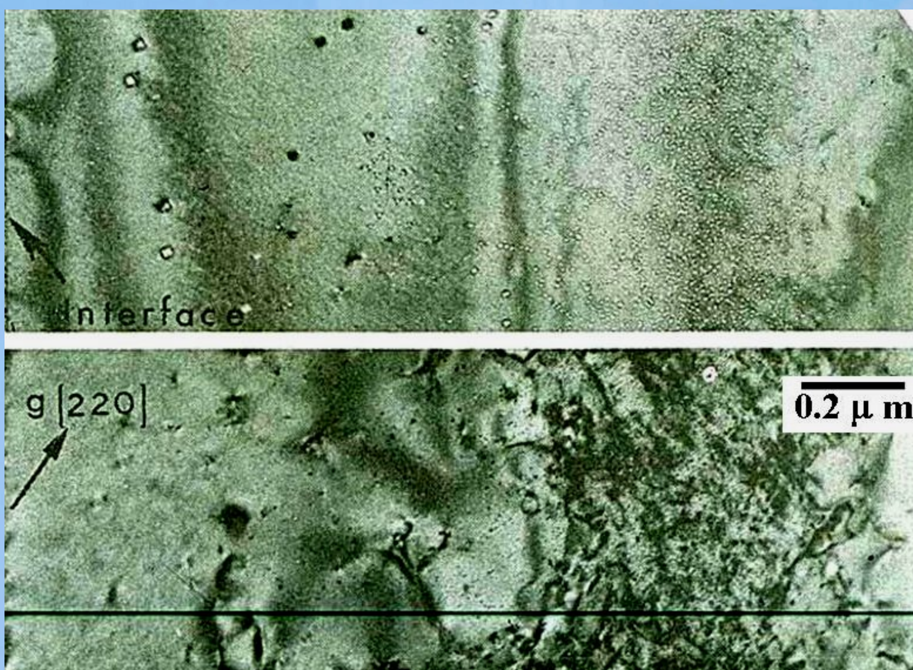

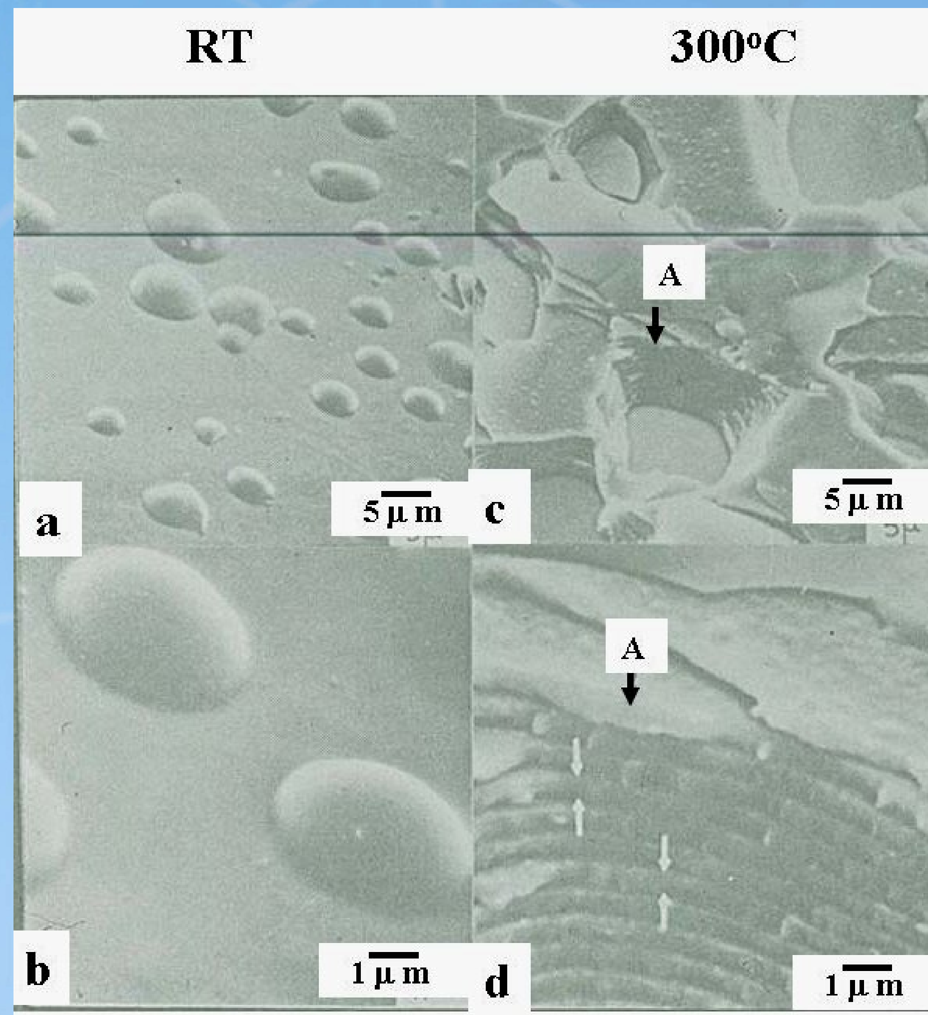

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Radiation Embrittlement

- **loss of fracture toughness in wrought and cast products**
- **continued research will test the irradiated samples from CIR, Halden, and Zorita to obtain Fracture Toughness data for irradiated samples that are representative of extended operating reactor life of 54 EFPY.**

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Thermal & Radiation Embrittlement

- **research work is planned**
  - → **to investigate the synergistic effect of thermal and radiation embrittlement on the *Fracture Toughness* of the Reactor Vessel Internals Equivalent to Reactor life 54 EFPY & Beyond**

Radiation Induced Stress Relaxation

- ➔ Investigate and Confirm the
  - ➤ Threshold Limits (at low dose) and the
  - ➤ Saturation Effects (at high dose)
    - – associated with the Loss of Preload on Fasteners Exposed to Neutron Fluence (Equivalent to Reactor life <u>54 EFPY</u> & <u>Beyond</u>)

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Proposed Use of Zorita Internals

- **IASCC in BWRS**
  - **Objectives**
    - Effect of HWC on IASCC Susceptibility and
    - CGR thresholds for IASCC onset and loss of HWC effectiveness
  - **Materials**
    - 304 SS baffle bolt and core shroud material, 308/309 weld material, and possibly 347 bolts
  - **Tests**
    - CGR, J-R, SSRT testing, and TEM

- **IASCC in PWRs**
  - **Objectives**
    - Threshold for onset of IASCC, ISACC susceptibility,
    - Effect of dose on mechanical properties such as strength, ductility and toughness, and
    - Dose dependence of irradiation hardening and embrittlement
  - **Materials**
    - 304 SS baffle bolt and core shroud material
  - **Tests**
    - CGR, J-R, SSRT, Ball Punch testing, and TEM

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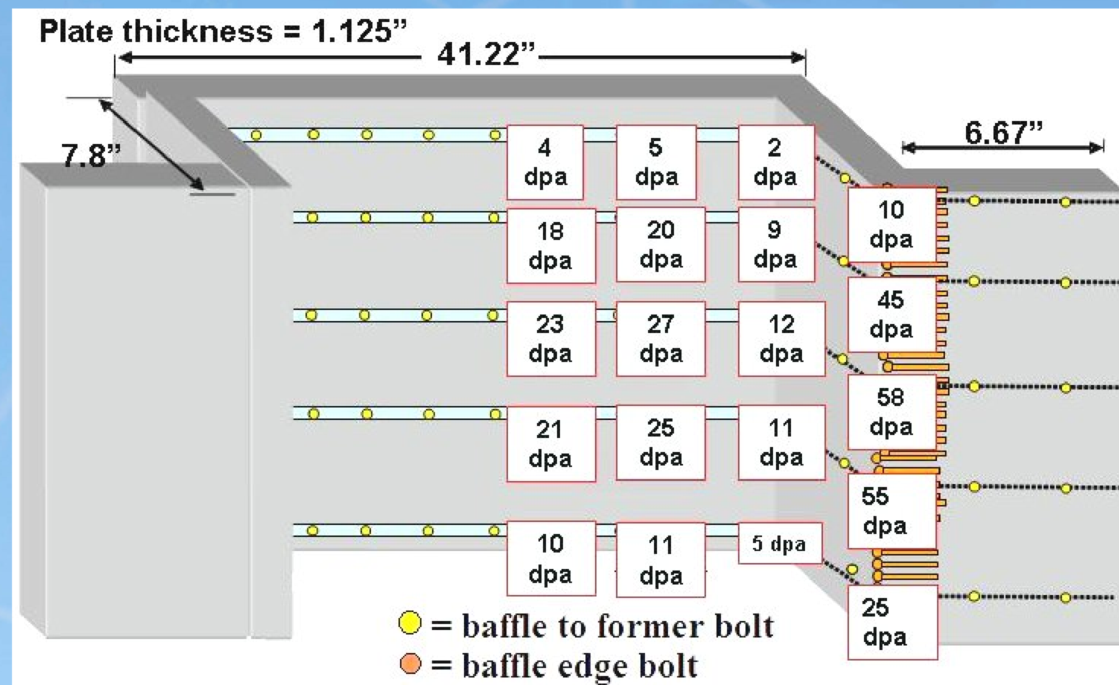
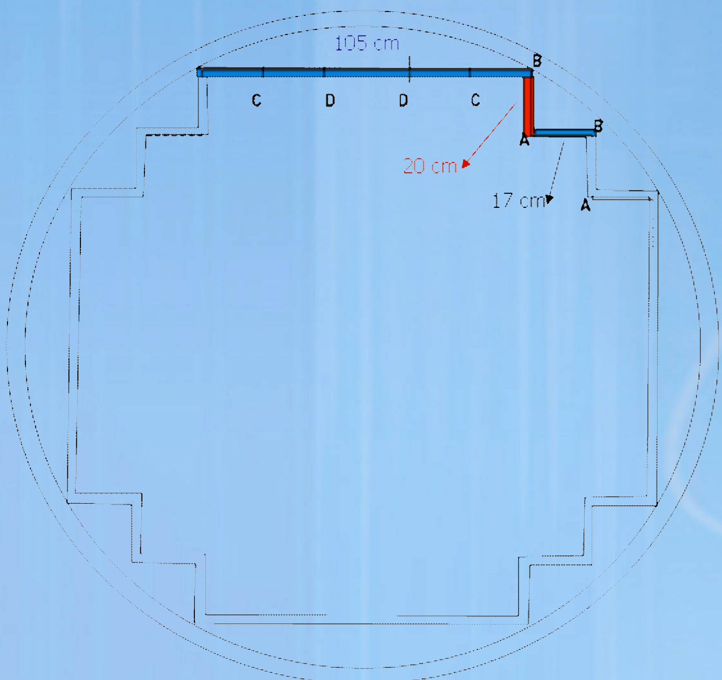


Fluence Levels of the Internals



Baffle Configuration

Core Internals Components

Form	Material	Thickness	Irradiation
Baffle plates	304 SS Annealed Hot Rolled & Pickled	≈ 2.85 cm	Up to 58 dpa
Baffle/former bolts	347 SS		Up to 58 dpa
Formers	304 SS Annealed Hot Rolled & Pickled	≈ 4 and 6 cm	
Core barrel	304 SS Annealed Hot Rolled & Pickled	≈ 4 cm	
Thermal shield	304		
Core barrel weld			1 dpa

Conclusion

- **Continue the Research Program to address IASCC in PWR and BWR environment**
  - **determine crack growth rates**
  - **determine threshold limits (at low dose)**
  - **saturation effects (at high dose) for the onset of IASCC due to the exposure to Neutron Fluences (Equivalent to Reactor life 54 EFPY & Beyond)**

Conclusion cont.

- Start Research Program to address
  - ➔ Void Swelling
    - due to the exposure to Neutron Fluences (Equivalent to Reactor life <u>54 EFPY</u> & <u>Beyond</u>)
  - ➔ Synergistic Effect of Thermal and Radiation Embrittlement
    - determine crack growth rates
  - ➔ Radiation Induced Stress Relaxation
    - determine threshold limits (at low dose)
    - saturation effects (at high dose)